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L-11-191

10 CFR 50.73

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT:
Beaver Valley Power Station, Unit No. 2
Docket No. 50-412, License No. NPF-73
LER 2011-002-00

Enclosed is Licensee Event Report (LER) 2011-002-00, "Auxiliary Feedwater System Vent Line Weld Crack Results in Technical Specification Required Plant Shutdown and Valid Reactor Protection System / Engineered Safety Feature Actuation System Actuations." This event is being reported in accordance with 10 CFR 50.73(a)(2)(i)(A), 10 CFR 50.73(a)(2)(iv)(A), 10 CFR 50.73(a)(2)(v)(B), and 10 CFR 50.73(a)(2)(v)(D).

There are no regulatory commitments contained in this submittal. Any actions discussed in this document that represent intended or planned actions are described for the NRC's information, and are not regulatory commitments.

If there are any questions or if additional information is required, please contact Mr. Brian T. Tuite, Manager, Regulatory Compliance at 724-682-4284.

Sincerely,

Paul A. Harden

Attachment

c: Mr. W. M. Dean, NRC Region I Administrator
Mr. D. L. Werkheiser, NRC Senior Resident Inspector
Ms. N. S. Morgan, NRR Project Manager
INPO Records Center (via electronic image)
Mr. L. E. Ryan (BRP/DEP)

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NRR

NRC FORM 366 (10-2010) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 10/31/2013
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 80 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA/Privacy Section (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects.resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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4. TITLE
Auxiliary Feedwater System Vent Line Weld Crack Results in Technical Specification Required Plant Shutdown and Valid Reactor Protection System / Engineered Safety Feature Actuation System Actuations

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									None	
04	09	2011	2011	- 002 -	00	06	03	2011	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
1	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
14 %	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☒ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(v)	☒ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☒ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER *(Include Area Code)*
Brian T. Tuite, Manager, Regulatory Compliance	(724) 682-4284

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX		CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	BA	VTV	A391	N						

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete EXPECTED SUBMISSION DATE).*	☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On April 9, 2011, Beaver Valley Power Station Unit No. 2 (BVPS-2) was in the process of power ascension near the end of the fifteenth refueling outage (2R15). BVPS-2 entered Mode 1 at 0931 hours. At approximately 2309 hours on April 9, 2011, a field operator identified an active leak on the 0.75 inch pipe nipple on vent valve 2FWE-940 located on the Auxiliary Feedwater System (AFW) "A" Steam Generator (SG) feedwater injection header. On April 9, 2011 at 2349 hours, the Shift Manager declared the "A" SG feedwater injection header inoperable due to the observed condition. Per Condition D of Technical Specification (TS) 3.7.5 "Auxiliary Feedwater (AFW) System", BVPS-2 was required to be in Mode 3 within six hours and be in Mode 4 within 18 hours. During the plant shutdown, level in "A" SG was approaching the predetermined low level manual trip criterion point of 25 percent and a manual reactor trip was performed. The manual reactor trip was successfully completed without complications. The Reactor Coolant System (RCS) cooldown resulted in a shrink of level in the "A" SG and the steam driven AFW pump successfully automatically started on low-low SG level in the "A" SG. AFW flow was injected into SGs "B" and "C". Since the "A" SG feedwater injection header had been previously isolated to repair the pipe nipple leak, AFW flow injection to the "A" SG did not occur. A manual main steam line isolation (SLI) was initiated to limit RCS cooldown. These events are reportable per: 10 CFR 50.73(a)(2)(i)(A) due to a completion of a plant shutdown required by Technical Specification, per 10 CFR 50.73(a)(2)(v)(B) and (D) as an event that could have prevented the fulfillment of a safety function for systems needed to remove decay heat and to mitigate the consequences of an accident, and per 10 CFR 50.73(a)(2)(iv)(B)(1) as a Reactor Protection System manual actuation (manual reactor trip), and also as a valid Engineered Safety Feature Actuation System (ESFAS) automatic actuation of the AFW system per 10 CFR 50.73(a)(2)(iv)(B)(6) and manual actuation of the SLI system per 10 CFR 50.73(a)(2)(iv)(B)(2). The root cause of the branch line crack was determined to be less than adequate design guidance contained in the design process to prevent vibration induced high cycle fatigue failure of the pipe nipple associated with 2FWE-940. The root cause of the manual reactor trip and automatic start of the steam driven AFW pump event was determined to be that the shutdown procedure guidance was not adequate for RCS temperature control and steam generator level control at the beginning of life conditions. The safety significance associated with these events is considered to be very low.

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There were no structures, components, or systems that were inoperable at the start of the event that contributed to the event. Energy Industry Identification System (EIIS) codes are identified in the text using the format [XX].

DESCRIPTION OF EVENT

On April 9, 2011, Beaver Valley Power Station Unit No. 2 (BVPS-2) was in the process of power ascension near the end of the fifteenth refueling outage (2R15). BVPS-2 entered Mode 1 at 0931 hours. At approximately 2309 hours on April 9, 2011, a field operator identified an active leak on the 0.75 inch pipe nipple on vent valve 2FWE-940 [VTV] which is on the Auxiliary Feedwater System (AFW) [BA] "A" Steam Generator (SG) feedwater injection header. This vent valve is located outside of containment on containment penetration number 79 between the credited outside containment penetration isolation valves (2FWE-HCV100E [HCV] and 2FWE-HCV100F [HCV]) and the outside containment wall penetration. The inside containment isolation barrier consists of a closed system (steam generator and associated connecting piping) which remained operable. Engineering and Operations personnel walk down of the leak on vent valve 2FWE-940, confirmed that it was a through wall crack at the weld toe of the 0.75 inch Class 2 branch line pipe-to-cupolet socket weld. A very small stream of water (approximately less than one-half of a gallon per minute) was spraying from the branch line crack.

On April 9, 2011 at 2349 hours, the BVPS-2 Shift Manager declared the "A" SG feedwater injection header inoperable due to the branch line crack. Per Condition D of Technical Specification (TS) 3.7.5 "Auxiliary Feedwater (AFW) System", BVPS-2 was required to be in Mode 3 within six hours (by 0549 hours on April 10, 2011) and be in Mode 4 within 18 hours (by 1749 hours on April 10, 2011). BVPS-2 also applied Condition C of TS 3.6.3 "Containment Isolation Valve" due to the branch line crack on 2FWE-940. The required actions of Condition C requires that the affected penetration flow path be isolated within 72 hours by use of at least one closed and de-activated automatic valve, closed manual valve or blind flange. TS 3.6.1 "Containment" Condition "A" was also entered due to the 2FWE-940 branch line crack. The "A" SG feedwater injection header remained available but not operable during this time frame.

At 0204 hours on April 10, 2011, the "A" SG feedwater injection header was isolated upstream of check valve 2FWE-99 [ISV] to repair the leak on 2FWE-940 and was no longer available. The cracked pipe nipple was replaced to the original configuration and the "A" SG feedwater injection header was returned to operable status on April 10, 2011 at 1339 hours.

On April 10, 2011, BVPS-2 was performing the TS 3.7.5 required shutdown as discussed above. At 0354 hours on April 10, 2011, BVPS-2 entered Mode 2 (i.e. less than five percent

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reactor power). The control room operations staff communicated that the level in "A" SG was approaching the predetermined SG low-low level manual reactor trip criterion point of 25 percent. At 0357 hours, "A" SG level reduced to 25 percent and a manual reactor trip was performed. The manual reactor trip was successfully completed without complications. All controls rods were fully inserted in the reactor core and safety systems and equipment functioned as expected. The control room operations staff entered Emergency Operating Procedure E-0, "Reactor Trip or Safety Injection", as expected. The Reactor Coolant System (RCS) [AB] cooldown resulted in a shrink of level in the "A" SG to the Engineered Safety Feature Actuation Signal (ESFAS) setpoint. The steam driven AFW pump successfully automatically started on low-low SG level in the "A" SG due to the ESFAS start signal. As a result of the AFW automatic start of the steam driven AFW pump, AFW flow was injected into the "B" and "C" SGs. AFW flow injection to the "A" SG did not occur since the "A" SG feedwater injection header had been previously isolated at 0204 hours to repair the 2FWE-940 branch line crack. The reactor protection system (RPS) logic for an automatic reactor trip was also satisfied due to a low-low water level in "A" SG. However, the safety function of the automatic reactor trip was previously accomplished when the manual reactor trip was completed. At 0400 hours, the control room operations staff transitioned from emergency operating procedure 2OM-53A.1.E-0 (Reactor Trip or Safety Injection) to 2OM-53A.1.ES-0.1 (Reactor Trip Response). The control room operations staff manually initiated main steam line isolation (SLI) utilizing the manual SLI pushbuttons in order to limit RCS cooldown per plant procedure 2OM-53A.1.ES-0.1. The SLI actuation includes closure of all three main steam line isolation valves (MSIVs). The manual SLI initiation was expected should a reactor trip occur at the beginning of core life and had been pre-briefed as part of the plant shutdown preparations. The manual SLI was completed successfully with the required components actuating as expected. In addition, AFW flow was reduced to limit the RCS cooldown. The control room operations staff performed 2OM-51.4.G, titled "Station Shutdown Establishing Mode 3 Conditions Following a Reactor Trip or SI", prior to returning to the normal shutdown procedure.

CAUSE OF EVENT

The most probable direct cause of the branch line crack was determined to be vibration induced high cycle fatigue failure of the vertical 0.75 inch pipe socket welded to the four inch "A" SG feedwater injection header line due to a less than adequate design in combination with a potential discontinuity in the socket weld toe.

The root cause of the branch line crack was determined to be less than adequate design guidance contained in the design process to prevent vibration induced high cycle fatigue failure of the pipe nipple associated with 2FWE-940. Various documents (i.e. Engineering Standards, Plant Installation Process Standards, and Weld Manual) did not contain guidance for design or fabrication that was available through industry recommendation documents relative to free-end branch line design. This AFW line vent line was installed by a plant modification performed in 1993.

The direct cause of the manual reactor trip and automatic start of the steam driven AFW pump was determined to be that during a beginning of life (BOL) reactor shutdown, the control room operations staff was not able to maintain adequate steam generator level. The RCS cooldown and SG shrink were greater than expected resulting in the control room operations staff initiating a manual reactor trip when narrow range level reduced to 25 percent in the "A" SG.

The root cause of the manual reactor trip and automatic start of the steam driven AFW pump was determined to be that the shutdown procedure guidance was not adequate for RCS temperature control and SG level control. The procedure for performing a controlled shutdown is the refueling outage shutdown procedure which was designed for end of life (EOL) conditions. Conditions at BOL are significantly different; however, the procedure had no guidance to address the difference in reactivity response from rod insertion or the expected SG level shrink. The station failed to recognize the need to develop procedure guidance for this scenario since controlled shutdowns at BOL are infrequent.

ANALYSIS OF EVENT

The AFW System automatically supplies feedwater to the SGs to remove decay heat from the RCS upon the loss of normal feedwater supply. The design basis of the AFW System is to supply water to the SG to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the SGs. In addition, the AFW System must supply enough makeup water to replace the SG secondary inventory lost as the unit cools to Mode 4 conditions. Sufficient AFW flow must also be available to account for flow losses such as pump recirculation and line breaks. The limiting Design Basis Accidents (DBAs) for the AFW System are loss of normal feedwater and feedwater line break. With one feedwater injection header inoperable, an insufficient number of SGs are available to meet the feedline break analysis. This analysis assumes AFW flow will be provided to the two remaining intact feedwater lines. Should a feedline break occur on one of the operable feedwater headers with one feedwater injection header already inoperable, the plant could no longer meet its safety analysis.

The branch line crack on the AFW vent line resulted in the "A" SG feedwater injection line being declared inoperable and the subsequent isolation of injection line to repair the branch line crack. When the line was isolated from 0204 hours until shortly before the

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feedwater injection header was restored to operable status at 1339 hours on April 10, 2011, this flow path was not available to supply AFW to the "A" SG. Following the discovery of the crack, the Technical Specification required actions were taken and a plant shutdown was initiated.

The safety significance associated with the BVPS-2 branch line crack on the piping associated with the AFW vent valve 2FWE-940 is considered to be very low. This is based on the delta core damage frequency and delta large early release frequency for the event during the limited period that the degraded condition existed.

The plant risk associated with the BVPS-2 manual reactor trip, the manual SLI, and automatic AFW actuation on April 10, 2011, due to a low water level condition in the "A" SG, while the plant was performing a Mode 1 to Mode 3 Technical Specification required shutdown is considered to be very low. This is based on the conditional core damage probability and conditional large early release probability for the event when considering the actual plant conditions that were present at the time of the event.

The branch line crack on the AFW vent line is reportable for the following reasons. The branch line crack resulted in the completion of a plant shutdown required by Technical Specification (TS), and is thus reportable per 10 CFR 50.73(a)(2)(i)(A). BVPS-2 completed a plant shutdown to Mode 3 in accordance with TS 3.7.5 Condition D at 0357 on April 10, 2011. The branch line crack also resulted in the common train "A" and "B" feedwater injection header to the "A" SG being inoperable, and is thus reportable per 10 CFR 50.73(a)(2)(v)(B) and 10 CFR 50.73(a)(2)(v)(D). With the common header inoperable and unavailable due to maintenance activities to repair the branch line crack, this configuration could have prevented the fulfillment of the safety function to remove residual heat (in the "A" SG) and to mitigate the consequences of an accident.

The manual reactor trip, the manual SLI, and the automatic start of the steam driven AFW pump is reportable per 10 CFR 50.73(a)(2)(iv)(A) as an event that was not part of a pre-planned sequence which resulted in the valid manual or automatic actuation of three systems listed in paragraph 10 CFR 50.73(a)(2)(iv)(B). The manual reactor trip was initiated due to the actual SG level on the "A" SG approaching the automatic trip setpoint. Therefore, this RPS manual actuation (Manual Reactor trip) is reportable per 10 CFR 50.73(a)(2)(iv)(B)(1). The manual SLI was initiated per plant procedure 2OM-53A.1.ES-0.1 (Reactor Trip Response) due to the continued cooldown of the RCS following the manual reactor trip. Therefore, this manual SLI actuation, which includes closure of multiple MSIVs, is reportable per 10 CFR 50.73(a)(2)(iv)(B)(2). The AFW automatic start of the steam driven AFW pump was initiated due to the actual SG level on "A" SG reaching the automatic start setpoint following the manual reactor trip. Therefore, this automatic actuation is reportable per 50.73(a)(2)(iv)(B)(6).

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This event was previously reported as pursuant to 10 CFR 50.72(b)(2)(i), 10 CFR 50.72(b)(2)(iv)(B), 10 CFR 50.72(b)(3)(ii)(A), and 10 CFR 50.72(b)(3)(iv)(A) at 0640 hours on April 10, 2011 (Event Notification (EN) Number 46744). However upon further review, the initial reportability of this event under 10 CFR 50.72(b)(3)(ii)(A) is retracted since the credited inside containment boundary (the steam generator closed system located inside containment) remained intact and operable during this event. The containment function or integrity was thus maintained during this event and therefore, this event did not result in serious degradation of a principal safety barrier. In addition, EN Number 46744 should have reported the manual SLI actuation under 10 CFR 50.72(b)(3)(iv)(A) as a valid system actuation. A loss of safety function should have also been reported under 10 CFR 50.72(b)(3)(v)(B) & (D) when the "A" feedwater injection header was declared inoperable and isolated for repair. This issue has been entered into the Corrective Action Program.

CORRECTIVE ACTIONS

Applicable to the AFW vent line branch line crack:

1. The cracked pipe nipple was replaced to the original configuration and the "A" SG feedwater injection header was returned to operable status on April 10, 2011 at 1339 hours. For the repair, the weld size and profile was increased (i.e. two to one leg length). This action should reduce stresses at the weld toe which should reduce the likelihood of a fatigue-type failure at this location.
2. Pipe supports are being designed to inhibit vibration on all three feedwater injection line vent lines.
3. Liquid penetrant testing of the branch line pipe-to-cupolet socket weld on the vent valves at the similar locations on the feedwater injection lines to the "B" and "C" Steam Generators will be performed.
4. A review against the Architectural Engineer (AE) vent/drain guidance document will be performed on select vents/drains located on systems as defined in the Extent of Cause Strategy.
5. Design guidance will be incorporated into engineering standards to address system induced vibration on free end branch connections.

Applicable to the manual reactor trip and AFW pump automatic start:

1. Develop procedure guidance for plant shutdown applicable to all times (beginning, middle, and end) of core life.

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2. BVPS-1 and BVPS-2 procedures governing plant startup and shutdown evolutions will be reviewed for content and appropriate guidance for expected initial conditions.

3. Additional simulator training on low power operation will be performed for BVPS-1 and BVPS-2 licensed operators as part of the continuing training program.

Completion of the above and other corrective actions are being tracked through the BVPS corrective action program.

PREVIOUS SIMILAR EVENTS

A review of previous BVPS LERs for approximately the previous five years determined that there were two LERs that involved similar events.

1. BVPS-1 LER 2010-002-01, "270 Degree Circumferential Flaw Found on Residual Heat Removal System Drain Valve Socket Weld."

Cause: Vibration induced fatigue, due to less than adequate design considerations

2. BVPS-2 LER 2008-001-00, "Unplanned Actuation of the Auxiliary Feedwater System During Plant Startup."

Cause: Unfamiliar with SG water level control during low power operation. Operations Management Team failed to ensure the startup crew was staged for success in operating the plant in a low power configuration.

CR 11-92597/11-92603